June 18, 2001

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT REQUEST FOR STEAM GENERATOR REPLACEMENT/POWER UPRATE - SHEARON HARRIS NUCLEAR POWER PLANT (TAC NOS. MB0199 AND MB0782)

Dear Mr. Scarola:

By letters dated October 4, and December 14, 2000, you requested license amendments to revise the Shearon Harris Nuclear Power Plant Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated core power level of 2900 MWt.

During the course of our review of these requests, the NRC staff has determined that additional information is necessary to complete our review. The enclosed request for additional information was e-mailed to your licensing staff on June 11, 2001, and discussed during a telephone call on June 14, 2001. A mutually agreeable target date of July 16, 2001, for your response was established. If circumstances result in the need to revise the target date, please call me at the earliest opportunity.

Sincerely,

/RA/

Richard J. Laufer, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure: As stated

cc w/encl: See next page

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Request for Additional Information
Request for License Amendment: Steam Generator Replacement/Power Uprate
Shearon Harris Nuclear Power Plant
Docket No. 50-400

1. In Section 2.3.2 of Reference 1, you indicated that the computer code used for the pipe stress analysis is different from that used in the original design basis analysis. Provide a justification that the new code was benchmarked for this application.

2. In Section 2.16.1.2-3 of Reference 1, you stated that the reactor coolant system (RCS) support loads on the internal concrete structures affect primarily the localized support areas. However, you did not discuss your evaluation of the local areas for increased pipe support loads. Provide a summary of the evaluation of local support areas for increased RCS support loads. If an evaluation was not performed, provide the basis for its exclusion.

3. In Section 2.16.1.2-4 of Reference 1, in evaluating three main steel platforms you stated that for qualifying internal steel structural platforms, base temperature of 60°F was used, although the effective base temperature is higher than 60°F. Provide the magnitude and justification for the higher effective base temperature used as a basis for qualifying internal steel structural platforms.

4. In Section 2.16.1.2-4 of Reference 1, with regard to the main steel platforms at elevation (EL) 236', 261', and 286', you stated that the governing load cases for the majority of critical steel member/connections do not include temperature load because these structures are generally free to expand under thermal loads due to slotted holes. Describe the method(s) you used to ensure that steel member/connections that are not free to expand under thermal loads have been evaluated for increased thermal loads in combination with the other design basis loads.

5. In Section 2.16.1.2-4 of Reference 1, with regard to the main steel platforms at EL 236', 261', and 286', you stated that higher allowable stresses are allowed for cases that include the accident temperature, and concluded that there is sufficient margin available to accommodate the potential increase of accident temperature in the main steel platforms due to the steam generator replacement/power uprate (SGR/Uprate). Provide the design basis margin and margins after considering increased accident temperature loads due to the SGR/Uprate.

6. In Section 2.16.1.2-4 of Reference 1, for main steam tower and main steam/feedwater hard restraint structures, and steam generator access platforms, provide evaluation summaries and design margins as a result of the power uprate.

7. In Section 2.16.2.2 of Reference 1, you stated that the existing peak accident pressure in the main steam tunnel is 6.47 psig. However, the Pre-SGR/Uprate main steam tunnel accident pressure provided in Table 2.23-2 is 18 psia (i.e., 3.3 psig). Discuss why the existing and Pre-SGR/UPrate main steam tunnel accident pressure is different in section 2.16.2.2 of reference 1 and in Table 2.23-2, respectively. Also, describe how the dynamic effects of the post SGR/Uprate accident pressure time history profile shown in Table 2.23-4 have been considered in the structural evaluation of the main steam tunnel, and internal steel structures and components.

8. In Section 2.16.2.2 of Reference 1, you stated that the increase in the maximum temperature in the main steam tunnel will affect the platform steel. Explain how the temperature increase in pipe support structural steel members that provide support to the platform steel was considered. You also concluded that structural failure of platform steel will not occur due to increase in main steel tunnel temperature and that the SGR/Uprate does not introduce seismic II/I concerns with these platforms. Provide a basis, including a description of analyses and evaluations, to justify your conclusions.

9. In Table 1-1 of Reference 2, you stated that for reactor internal components evaluation one of the computer codes is different from those used in the original design basis analysis. Provide a justification that the new code was benchmarked for this application.

10. In Section 5.1.1.2 of Reference 2, with regard to the evaluation of the core support pads, you stated that the combined normal plusloss-of-coolant accident (LOCA) stresses were compared to the applicable faulted condition acceptance criteria. Provide a justification for not combining stresses due to safe shutdown earthquake with LOCA stresses in the faulted load combination.

11. In Table 5.1.1-1 of Reference 2, you stated that the maximum range of stress intensity for reactor vessel closure studs, 97.5 ksi, is less than the code-allowable stress of 80.1 Ksi. This statement appears to be in error. Provide a justification for the adequacy of the reactor vessel closure studs.

12. In Section 5.5.1.3.3 of Reference 2, with regard to the steam generator displacements, discuss how the steam generator displacements were addressed in evaluating the steam generator attached piping.

13. In Section 5.6.1.2 of Reference 2, with regard to fatigue analysis of the reactor coolant pump, you indicated that none of the changes to the normal or upset transients for the SGR/Uprating cause a non-significant pressure or thermal transient to become a significant transient. Discuss the criteria used in determining the significance of pressure and thermal transients for the fatigue analysis of the reactor coolant pump.

14. In Section 5.8 of Reference 2, with regard to the pressurizer, discuss the evaluations performed for the pressurizer safety valves and the power-operated relief valves.

15. Confirm that safety-related motor-operated valves will be capable of performing their intended function(s) following the SGR/Uprate, including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the SGR/Uprated power level was not evaluated.

16. Clarify whether you have evaluated the effect of increased temperature due to power uprate on thermally induced pressurization of piping runs penetrating the containment that were evaluated in response to Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions.”

17. In Reference 3, with regard to the impact of power uprate on the spent fuel pool (SFP) cooling and cleanup system, you stated that the uprate analyses have been performed by revising the single active failure assumption to be a loss of just a single SFP cooling pump. Provide a justification for the deviation from the single active failure assumption described in the FSAR that assumes the loss of one of the two cooling trains for the SFP (a pump and heat exchanger). Also, for SFP conditions concurrent with a design basis LOCA, provide the maximum calculated SFP temperature.

REFERENCES

(1) Balance of Plant (BOP) licensing report, enclosure 7 to Serial: HNP-00-142, dated October 4, 2000

(2) WCAP-15398, Westinghouse Proprietary Class 2C, NSSS licensing report, enclosure 8 to Serial: HNP-00-142, dated October 4, 2000

(3) Enclosure 6 to Serial: HNP-00-175, dated December 14, 2001